10CFR50.73

Virginia Electric and Power Company
North Anna Power Station
P. O. Box 402
Mineral, Virginia 23117

January 9, 2007

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555-0001

Serial No.: 06-1018
NAPS: MPW
Docket No.: 50-339
License No.: NPF-7

Dear Sirs:

Pursuant to 10CFR50.73, Virginia Electric and Power Company hereby submits the following Licensee Event Report applicable to North Anna Power Station Unit 2.

Report No. 50-339/2006-001-00

This report has been reviewed by the Station Nuclear Safety and Operating Committee and will be forwarded to the Management Safety Review Committee for its review.

Sincerely,

Dan Stoddard

Daniel G. Stoddard
Site Vice President
North Anna Power Station

Enclosure

Commitments contained in this letter: None

cc: United States Nuclear Regulatory Commission
Region II
Sam Nunn Atlanta Federal Center
61 Forsyth Street, SW, Suite 23T85
Atlanta, Georgia 30303-8931

Mr. J. T. Reece
NRC Senior Resident Inspector
North Anna Power Station

IE22

NRC FORM 366 U.S. NUCLEAR REGULATORY COMMISSION
(6-2004)

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
NORTH ANNA POWER STATION , UNIT 2	05000 339	1 OF 4

4. TITLE
Reactor Trip Due To Steam Generator Low Level Coincident With A Steam Flow Feed Flow Mismatch

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCUMENT NUMBER 05000
11	16	2006	2006	-- 001 --	00	01	09	2007	FACILITY NAME	DOCUMENT NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20.2201(b)		20.2203(a)(3)(ii)		50.73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50.73(a)(2)(iii)		50.73(a)(2)(x)
	20.2203(a)(1)		50.36(c)(1)(i)(A)	X	50.73(a)(2)(iv)(A)		73.71(a)(4)
	20.2203(a)(2)(i)		50.36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73.71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50.73(a)(2)(v)(B)		OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50.73(a)(2)(i)(B)		50.73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50.73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50.73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER (Include Area Code)
D. G. Stoddard, Site Vice President	(540) 894-2101

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIX
X	SJ	FCV	W120	Y					

14. SUPPLEMENTAL REPORT EXPECTED		15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
YES (If yes, complete 15. EXPECTED SUBMISSION DATE)	X NO				

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On November 16, 2006, at 0226 hours with Unit 2 operating at 100 percent power an automatic reactor trip occurred. The initiating signal was the "B" steam generator (SG) low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "B" main feed regulating valve. This resulted in a reactor and turbine trip. Closure of the "B" main feed regulating valve was due to failure of the isolation card that provides steam flow input to the “B” SG water level control circuit. At 0357 hours a 4 hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iv)(B). An 8 hour Non-Emergency Report was also made to the NRC in accordance with 10 CFR 50.72 (b)(3)(iv)(A). This event is reportable pursuant to 10 CFR 50.73 (a)(2)(iv)(A) for a condition that resulted in an automatic actuation of any engineered safety feature including the reactor protection system. This event posed no significant safety implications because the Reactor Protection System and Engineered Safety Features Actuation Systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
NORTH ANNA POWER STATION UNIT 2	05000 - 339	2006	--001 --	00	2 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

1.0 DESCRIPTION OF THE EVENT

On November 16, 2006, at 0226 hours with Unit 2 operating at 100 percent power an automatic reactor trip occurred. The initiating signal was the "B" steam generator (SG) (EIIS System AB, Component SG) low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "B" main feed regulating valve (MFRV) (EIIS System SJ, Component FCV). This resulted in a reactor and turbine trip.

Control Room personnel responded to the event in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Control Room personnel stabilized the plant using ES-0.1 Reactor Trip recovery. Initially, Reactor Coolant System (RCS) (EIIS System AB) pressure and temperature decreased to approximately 1950 psig and 544 degrees Fahrenheit. Subsequently, RCS pressure and temperature returned to their normal programmed values.

Following the reactor trip the Reactor Protection System (RPS) and all Engineered Safety Feature Actuation System (ESFAS) (EIIS System JE) equipment responded as designed including proper operation of AMSAC, and the Auxiliary Feedwater System (AFW) (EIIS System BA). No other major equipment issues were noted.

At 0357 hours a 4 hour Non-Emergency Report was made to the NRC in accordance with 10 CFR 50.72 (b)(2)(iv)(B) for an event causing actuation of the Reactor Protection System when the reactor is critical. An 8 hour Non-Emergency Report was also made to the NRC in accordance with 10 CFR 50.72 (b)(3)(iv)(A) for an event causing actuation of the Auxiliary Feedwater System.

2.0 SIGNIFICANT SAFETY CONSEQUENCES AND IMPLICATIONS

This event posed no significant safety implications because the RPS and ESFAS systems functioned as designed following the reactor trip. Therefore, the health and safety of the public were not affected by this event.

This event is reportable pursuant to 10 CFR 50.73 (a)(2)(iv)(A) for a condition that resulted in an automatic actuation of any engineered safety feature including the reactor protection system.

3.0 CAUSE

Cause of the automatic reactor trip was the "B" SG low level coincident with a steam flow greater than feed flow mismatch. The initiating signal was caused by closure of the "B" MFRV. Closure of the "B" MFRV was the result of a failed isolator card that provides steam flow input to the "B" SG water level control circuit. The isolator card failure (i.e., de-energized) was determined to be a failure of one or more transistors in the power supply

FACILITY NAME (1)	DOCKET	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
NORTH ANNA POWER STATION UNIT 2	05000 - 339	2006	--001 --	00	3 OF 4

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

circuit of the card. The root cause of the transistor failure is age-related degradation.

Investigation determined that the failed card was installed in 1977 and had only been periodically calibrated and not undergone any repair or refurbishment during that time. The improvements in failure rate since 1993 combined with the lack of a clear industry standard or Westinghouse Guideline for 7300 System Printed Circuit Board (PCB) expected life led to PCBs remaining in service without replacement or refurbishment unless a failure occurred.

The extent of condition applies to those 7300 system PCBs that have exceeded the 15 year recommended life expectancy stated in the latest draft (dated Dec. 2006) of the 7300 System Life Cycle Management Planning Sourcebook. The extent of cause applies to PCBs in all systems, which currently do not have a replacement/refurbishment strategy.

A review of history shows no other failures of this isolator card have occurred and therefore this event is not considered a repeat. A reactor trip of Unit 2 that occurred in March 2003 was attributed to a fuse failure in the driver card, which caused the “C” Main Feedwater Control Valve to close. The corrective actions from that event focused solely on the driver cards. At that time fuses were inspected on both units with repairs made to several cards.

4.0 IMMEDIATE CORRECTIVE ACTION(S)

Control Room personnel responded to the event in accordance with emergency procedure E-0, Reactor Trip or Safety Injection. Control Room personnel stabilized the plant using ES-0.1 Reactor Trip Recovery. All safety systems responded appropriately. The unit was stabilized at no-load conditions, the Main Feedwater System was placed in service to all three S/Gs and the AFW System secured and returned to normal AUTO/Standby alignment. Subsequently, Control Room personnel transitioned to 2-OP-1.5 in preparation for unit re-start.

5.0 ADDITIONAL CORRECTIVE ACTIONS

The "B" MFRV failed isolator card was replaced and a successful functional test was performed. Unit 2 entered Mode 1 at 0336 hours on November 17, 2006. Unit 2 achieved 100 percent power at 0143 hours on November 18, 2006.

6.0 ACTIONS TO PREVENT RECURRENCE

The root cause evaluation determined a need to develop a replacement/refurbishment strategy for 7300 System PCBs in “critical” control loops, including work scope for the 2007 refueling outages, for PCBs that are 15 years or older. Also, to develop a systematic replacement/refurbishment strategy for the remaining 7300 PCB applications. Preventive Maintenance procedures are being established to ensure reliable system performance of

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* **(17)**

the 7300 Process Protection & Control System equipment based on Industry Best Practices.

7.0 SIMILAR EVENTS

LER N2-03-001-00 dated 03/31/03, documents an automatic reactor trip from "C" steam generator low level coincident with a steam flow greater than feed flow mismatch caused by closure of the "C" MFRV. Closure of the "C" MFRV was the result of a failed driver card in the SG water level control system for "C" SG. The driver card failed as a result of a blown fuse. The corrective actions from this event focused solely on the driver cards. Fuses were inspected on both units with repairs made to several cards.

8.0 ADDITIONAL INFORMATION

At the time of this event Unit 1 was in Mode 3 preparing for re-start following a mid-cycle outage.

Component information:
Description: Isolator Card Mark No. 02-MS-FM-2484A
Manufacturer: Westinghouse W893
Model No.: 2837A12G03
Serial No.: 89489